Mr. James A. Gresham, Manager June 2, 2005
Regulatory and Licensing Engineering
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: DRAFT SAFETY EVALUATION FOR TOPICAL REPORT WCAP-15836-P, "FUEL ROD DESIGN METHODS FOR BOILING WATER REACTORS - SUPPLEMENT 1" (TAC NO. MB5740)

Dear Mr. Gresham:

By letter dated June 25, 2002, as supplemented by letters dated April 16, and July 30, 2004, March 9, and April 22, 2005, Westinghouse Electric Company (Westinghouse) requested review and approval of topical report (TR) WCAP-15836-P, "Fuel Rod Design Methods for Boiling Water Reactors - Supplement 1." Enclosed for Westinghouse review and comment is a copy of the staff's draft safety evaluation (SE) for the TR.

Pursuant to 10 CFR 2.390, we have determined that the enclosed draft SE does not contain proprietary information. However, we will delay placing the draft SE in the public document room for a period of ten working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects. If you believe that any information in the enclosure is proprietary, please identify such information line-by-line and define the basis pursuant to the criteria of 10 CFR 2.390. We intend to treat the entire Technical Evaluation Report as proprietary and not make it publicly available. After ten working days, the draft SE will be made publicly available, and an additional ten working days are provided to you to comment on any factual errors or clarity concerns contained in the SE. The final SE will be issued after making any necessary changes and will be made publicly available. The staff's disposition of your comments on the draft SE will be discussed in the final SE.

To facilitate the NRC staff's review of your comments, please provide a marked-up copy of the draft SE showing proposed changes and provide a summary table of the proposed changes.

If you have any questions, please contact Brian Benney at (301) 415-3764.

Sincerely,

/RA/
Robert A. Gramm, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 700

Enclosures: 1. Draft Safety Evaluation
2. Technical Evaluation Report **(Proprietary)**

cc w/encls: See next page

cc:
Mr. Gordon Bischoff, Manager
Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

June 2, 2005

Mr. James A. Gresham, Manager
Regulatory and Licensing Engineering
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: DRAFT SAFETY EVALUATION FOR TOPICAL REPORT WCAP-15836-P, "FUEL ROD DESIGN METHODS FOR BOILING WATER REACTORS - SUPPLEMENT 1" (TAC NO. MB5740)

Dear Mr. Gresham:

By letter dated June 25, 2002, as supplemented by letters dated April 16, and July 30, 2004, March 9, and April 22, 2005, Westinghouse Electric Company (Westinghouse) requested review and approval of topical report (TR) WCAP-15836-P, "Fuel Rod Design Methods for Boiling Water Reactors - Supplement 1." Enclosed for Westinghouse review and comment is a copy of the staff's draft safety evaluation (SE) for the TR.

Pursuant to 10 CFR 2.390, we have determined that the enclosed draft SE does not contain proprietary information. However, we will delay placing the draft SE in the public document room for a period of ten working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects. If you believe that any information in the enclosure is proprietary, please identify such information line-by-line and define the basis pursuant to the criteria of 10 CFR 2.390. We intend to treat the entire Technical Evaluation Report as proprietary and not make it publicly available. After ten working days, the draft SE will be made publicly available, and an additional ten working days are provided to you to comment on any factual errors or clarity concerns contained in the SE. The final SE will be issued after making any necessary changes and will be made publicly available. The staff's disposition of your comments on the draft SE will be discussed in the final SE.

To facilitate the NRC staff's review of your comments, please provide a marked-up copy of the draft SE showing proposed changes and provide a summary table of the proposed changes.

If you have any questions, please contact Brian Benney at (301) 415-3764.

Sincerely,
/RA/
Robert A. Gramm, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 700

Enclosures: 1. Draft Safety Evaluation
2. Technical Evaluation Report **(Proprietary)**

cc w/encls: See next page

DISTRIBUTION:
PUBLIC **(No DPC for 10 working days)**
PDIV-2 Reading
RidsNrrDlpmLpdiv (HBerkow)
RidsNrrDlpmLpdiv-2 (RGramm)
RidsNrrPMBBenney
RidsAcrsAcnwMailCenter
RidsOgcRp
FAkstulewicz
PClifford
RidsNrrLALFeizollahi

ADAMS Package: ML051540102
ADAMS Letter Accession No.: ML051530526
Proprietary Technical Report (Non-Public): ML051540158
NRR-106

OFFICE	PDIV-2/PM	PDIV-2/LA	SRXB-A/SC	PDIV-2/SC	PDIV-2/D
NAME	BBenney	LFeizollahi	FAkstulewicz	RGramm	HBerkow
DATE	6/2/05	6/1/05	5/26/05	5/31/05	6/2/05

E:\Filenet\ML051530526.wpd

OFFICIAL RECORD COPY

DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT WCAP-15836-P, "FUEL ROD DESIGN METHODS FOR BOILING WATER REACTORS - SUPPLEMENT 1"

WESTINGHOUSE ELECTRIC COMPANY LLC

1.0 INTRODUCTION

By letter dated June 25, 2002 (Reference 1), as supplemented by letters dated April 16, 2004 (Reference 2), July 30, 2004 (Reference 3), March 9, 2005 (Reference 4), and April 22, 2005 (Reference 5), Westinghouse Electric Company (Westinghouse) requested review and approval of WCAP-15836-P, entitled, "Fuel Rod Design Methods for Boiling Water Reactors - Supplement 1." This licensing topical report (TR) describes improvements to the previously approved Boiling Water Reactor (BWR) fuel performance codes STAV, VIK, and COLLAPS (Reference 6). The new code versions, STAV7.2, VIK-3, and COLLAPS II, Version 3.3D, are intended to support fuel design and licensing applications up to a rod average burnup of 62 GWd/MTU.

The Nuclear Regulatory Commission (NRC) staff's review was assisted by Pacific Northwest National Laboratory (PNNL). The NRC staff's conclusions on the acceptability of WCAP-15836-P are supported by the proprietary PNNL Technical Evaluation Report (TER) which is being withheld from public availability.

2.0 REGULATORY EVALUATION

Regulatory guidance for the review of fuel system designs and adherence to applicable General Design Criteria (GDC) is provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 4.2, "Fuel System Design." In accordance with SRP Section 4.2, the objectives of the fuel system safety review are to provide assurance that:

a. The fuel system is not damaged as a result of normal operation and anticipated operational occurrences (AOOs),
b. Fuel system damage is never so severe as to prevent control rod insertion when it is required,
c. The number of fuel rod failures is not underestimated for postulated accidents, and
d. Coolability is always maintained.

In addition to licensed reload methodologies, fuel performance models are utilized, along with an approved mechanical design methodology, to demonstrate compliance to SRP Section 4.2 fuel design criteria. WCAP-15836-P describes improvements to Westinghouse's suite of BWR fuel performance computer models (STAV7.2, VIK-3, and COLLAPS II, Version 3.3D). The

NRC staff's review of WCAP-15836-P is to ensure that these computer models are capable of accurately (or conservatively) predicting the in-reactor performance of fuel rods and to identify any limitations on the ability of the code to perform this task. A subsequent TR on fuel mechanical design methodology (WCAP-15942-P) will discuss how a Westinghouse BWR fuel design, employing these models, demonstrates compliance to the applicable regulatory requirements identified in SRP Section 4.2.

3.0 TECHNICAL EVALUATION

The NRC staff's review of these fuel performance models is summarized below:

- Verify material properties based on supporting mechanical testing database.

- Verify each model (e.g., fuel temperature, creep, etc.) based on separate effects testing and measurements.

- Verify synergistic interaction of coupled models based on comparisons to instrumented in-pile test programs.

- Verify predicted in-reactor performance based on pool-side and hot-cell irradiation database.

In addition to comparing the computer model predictions to the supporting database, PNNL performed extensive bench marking of STAV7.2 against the NRC audit code FRAPCON-3. The fuel performance models in FRAPCON-3 have been validated against an extensive database and are continually assessed against newer data as it becomes available.

3.1 Thermal Modeling

Pellet Heat Generation and Heat Transfer Methods
The solution method for the heat generation within the pellet has been improved in STAV7.2. Fuel and cladding temperatures are calculated assuming steady-state, radial-only heat transfer from the pellet, across the pellet-cladding gap, through the cladding base metal, across the oxide and crud layers, and across the water film to the coolant. The PNNL technical assessment of the heat generation and heat transfer solution methods is provided in Section 2.1 of the supporting TER. Based upon this assessment, the NRC staff finds the pellet heat generation and heat transfer solution methods in STAV7.2 acceptable.

Fuel Thermal Conductivity
Based upon FRAPCON-3 benchmarks and comparisons to relevant empirical data, PNNL concluded that the STAV7.2 UO_2 pellet thermal conductivity model was non-conservative (Section 2.2 of supporting TER). In response (Request for Additional Information (RAI) No. 1 and No. 2, Reference 3), Westinghouse modified the pellet thermal conductivity model. Westinghouse subsequently re-performed the validation cases which were then benchmarked by PNNL to FRAPCON-3 and compared to relevant data. Figures 2.2.1 through 2.2.4 of the supporting TER illustrate these comparisons. Based upon these latest comparisons, PNNL concluded that the revised STAV7.2 UO_2 pellet thermal conductivity model was acceptable. Based upon this assessment, the NRC staff finds the revised UO_2 pellet thermal conductivity model acceptable.

Because the nominal fuel density of the in-reactor fuel database used to validate the STAV7.2 fuel thermal conductivity model is between 92 percent theoretical density and 97 percent theoretical density, the applicability of STAV7.2 will be limited to within this range.

Incorporating an empirical urania-gadolinia thermal conductivity correlation into the FRAPCON-3 model, PNNL compared benchmark cases to the revised STAV7.2 thermal conductivity model (Section 2.2.1 of supporting TER). PNNL concluded that the STAV7.2 thermal conductivity adjustment for gadolinia is acceptable and applicable up to the 9.0 wt% gadolinia concentration requested by Westinghouse. Based upon this assessment, the NRC staff finds the revised UO_2-Gd_2O_3 pellet thermal conductivity model acceptable and applicable up to 9.0 wt% gadolinia.

Westinghouse has not requested approval of STAV7.2 for fuel pellets containing additives other than gadolinia, nor has thermal conductivity data for such additives been provided. As such, approval for STAV7.2 will be limited to UO_2 and UO_2-Gd_2O_3 fuel pellets with no additives beyond nominal trace elements.

Gap Conductivity

Based upon comments received from PNNL, Westinghouse revised their pellet-cladding gap thermal model (Attachment 1 of Reference 3). The revised model was compared to FRAPCON-3 and found to be acceptably conservative. In addition, the accommodation coefficients for the gap gas species has been updated in STAV7.2. The values of these gas coefficients are identical to values in MATPRO. Based upon this assessment, the NRC staff finds the pellet-cladding gap conductivity model acceptable.

Fuel Thermal Expansion

The fuel thermal expansion model in STAV7.2 has not changed relative to STAV6.2. PNNL conducted a benchmark against FRAPCON-3 (Section 2.4 of supporting TER) and identified that the STAV7.2 code does not model the large increase in fuel volume during fuel melting. Because the code does not model this known phenomenon, its applications will be limited to fuel temperatures less than the melting temperature. Based on this assessment, the NRC staff finds the fuel thermal expansion acceptable with the condition that the use of STAV7.2 is limited to applications where the fuel temperature remains below the melting temperature.

Fuel Relocation

The overall gap thermal conductance model, which is influenced by fuel pellet radial relocation (due to thermal cracking and outward movement), was revised by Westinghouse (Attachment 1 of Reference 3) in response to concerns raised by PNNL (Section 2.5 of supporting TER). In conjunction, Westinghouse modified the fuel pellet relocation model. The revised relocation model was benchmarked to FRAPCON-3 and found to be in good agreement (Figures 2.5.1 and 2.5.2 of supporting TER). Based upon this assessment, the NRC staff finds the revised fuel pellet relocation model acceptable.

Clad-to-Coolant Heat Transfer Model
The clad-to-coolant heat transfer model in STAV7.2 is unchanged from STAV6.2. PNNL compared this model to the one used by FRAPCON-3 and concluded that it was acceptable. The clad-to-coolant heat transfer model is not burnup dependent and continues to be applicable.

Clad Thermal Conductivity
Over its range of applicability, the STAV7.2 clad thermal conductivity was compared to the model in FRAPCON-3 (Figure 2.6.2 of supporting TER). PNNL concluded that the STAV7.2 oxide conductivity was acceptable. Based upon this assessment, the NRC staff finds the clad thermal conductivity acceptable.

Clad Oxide Thermal Conductivity
Over its range of applicability, the STAV7.2 clad oxide thermal conductivity was compared to the model in FRAPCON-3 (Figure 2.6.1 of supporting TER). PNNL concluded that the STAV7.2 oxide conductivity was acceptable. Based upon this assessment, the NRC staff finds the clad oxide thermal conductivity acceptable.

Crud Thermal Conductivity
PNNL compared the crud conductivity value for BWRs in STAV7.2 against the conductivity assigned in FRAPCON-3 (Section 2.6 of the supporting TER). STAV7.2 utilizes a lower conductivity which will promote a greater temperature drop across the crud layer and conservatively higher fuel and clad temperatures. Based on this, PNNL concluded that the crud conductivity was conservative. The NRC staff agrees with this assessment and finds the crud conductivity value acceptable.

STAV7.2 Thermal Model Integral Assessment
Section 3.2 of WCAP-15836-P (as updated by Attachment 2 of Reference 3) describes the calibration and verification of the STAV7.2 thermal model. The STAV7.2 beginning of life (BOL) fuel centerline temperatures compare well to measured test rods from Halden. In-life fuel temperature data from several Halden test rods were also compared against STAV7.2 temperature predictions and found to be in good agreement. As part of the calibration process in STAV7.2, tuning parameters in the code algorithms are adjusted to achieve a best-estimate fit to the empirical database. As such, the values of these tuning parameters become an inherent part of the approved model. The PNNL integral assessment of the thermal models is discussed in Section 2.7 of the supporting TER. Based upon the Halden validation (comparisons of predicted fuel temperature versus the Halden test database) and the PNNL technical assessment, the NRC staff finds the overall interaction of the coupled heat generation and heat transfer models acceptable.

3.2 Fission Gas Release Model

The fission gas release model in STAV7.2 is a two-stage diffusion model that simulates the diffusion of gas through the grain to the grain boundary and the release from the grain boundary to the void volume. The diffusion constant in STAV7.2 is modified for the effect of gadolinia. The PNNL technical assessment of the fission gas release model is discussed in Section 3.1 of the supporting TER. In their review, PNNL requested further justification of the diffusion constant for UO_2-Gd_2O_3 fuel. In response (RAI No. 14 of Reference 2), Westinghouse referred to empirical data obtained from gamma scans which target certain mobile and immobile fission nuclides. These gamma scans demonstrate the diffusion of fission products (via tracking mobile nuclides) as well as characterize the burnup (via examining the immobile nuclides) for both UO_2 and UO_2-Gd_2O_3 fuel. The applicability of the diffusion coefficient and supporting gamma scan data for UO_2-Gd_2O_3 fuel up to the requested gadolinia concentration of 9.0 wt% was questioned. In response (RAI No. 9 of Reference 4), Westinghouse agreed to modify the diffusion model to saturate the effects of gadolinia at the limit of the empirical database.

PNNL compared the STAV7.2 fission gas release predictions to FRAPCON-3. While the algorithms and tuning parameters differ, the two codes have reasonably good agreement. In most cases, STAV7.2 predicts greater fission gas release which promotes conservative fuel rod internal pressure and fuel temperature calculations.

Grain Growth

PNNL compared the grain growth model in STAV7.2 against an internationally acknowledged model and empirical data (Section 3.2 of supporting TER). This comparison revealed that STAV7.2 over-predicts grain size which would tend to under-predict fission gas release.

STAV7.2 Fission Gas Release Integral Assessment

Section 3.3 of WCAP-15836-P (as updated by Attachment 2 of Reference 3) describes the calibration and verification of the STAV7.2 fission gas release models. As part of the calibration process in STAV7.2, tuning parameters in both the thermal and athermal code algorithms are adjusted to achieve a best-estimate fit (and in some cases a 95 percent upper-bound) to the empirical database. As such, the value of these tuning parameters become an inherent part of the approved model. The PNNL integral assessment of the fission gas release models is discussed in Section 3.3 of the supporting TER.

The Westinghouse database utilized in the calibration and verification process of the steady-state fission gas release models consisted of 261 BWR rods and 130 pressurized water reactor (PWR) rods. The transient fission gas database consisted of power ramp data from 60 BWR and PWR fuel rods conducted in test reactors (e.g. Studsvik). Appendix C of WCAP-15836-P characterizes the fuel rod database. The UO_2-Gd_2O_3 fuel database consists of 15 BWR rods (steady-state) and 3 PWR rods (ramp tested). Although the STAV7.2 code will not be used for PWR fuel rods, the calibration of the model to the expanded PWR database provides a greater degree of certainty to the model, especially at higher burnup.

Based on a comparison of the STAV7.2 predictions to the measured test data coupled with comparisons with FRAPCON-3, PNNL concluded that the STAV7.2 fission gas release models, with the revised gadolinia diffusion coefficient, is acceptable up to a burnup of 62 GWd/MTU. Although the grain growth model appears non-conservative, its impact on the predicted fission gas release is counteracted by the empirical nature of the calibration process. Based upon the calibration and verification exercise presented in Section 3.3 of WCAP-15836-P (as updated by Attachment 2 of Reference 3) and the PNNL technical assessment, the NRC staff finds the steady-state and transient fission gas release models acceptable up to a peak rod average burnup of 62 GWd/MTU.

3.3 Cladding Corrosion and Crud Models

A model which accurately predicts cladding waterside corrosion and crud deposition is required because the development of a corrosion layer and/or crud layer promotes increased thermal resistance. This increase in thermal resistance may result in higher fuel and cladding temperatures. In addition, the formation of hydrides may result in a reduction of cladding ductility.

The oxidation rate in STAV7.2 for a BWR is a function of both time and linear heat generation rate (LHGR). The overall corrosion rate is the sum of the nodular corrosion rate (athermal) and the diffusion controlled rate (strong function of temperature). Since corrosion rate is temperature dependent, buildup of a crud layer (which promotes higher clad temperatures) will result in an increase in oxidation rate.

<u>Oxidation Rate</u>
PNNL benchmarked STAV7.2 against the BWR oxidation model in FRAPCON-3. An examination of Figure 4.1.1 of the supporting TER reveals that STAV7.2 predicts a significantly larger amount of oxidation for a typical 10x10 BWR fuel design. This conservative oxidation prediction would promote a larger temperature change across the cladding and higher fuel temperatures.

<u>Crud Deposition</u>
PNNL investigated the crud deposition model in STAV7.2. Figure 4.1.2 of the supporting TER illustrates crud buildup as a function of burnup for two operating power histories. Also shown on this figure is the impact of the crud layer on temperature rise across the crud layer. PNNL benchmarked STAV7.2 against the BWR oxidation model in FRAPCON-3 to investigate the effect of crud deposition on oxidation rate. Figure 4.1.3 of the supporting TER illustrates this effect as well as demonstrating that the combined crud and corrosion models in STAV7.2 are conservative relative to FRAPCON-3.

<u>Hydrogen Pickup</u>
The hydriding model is a stand-alone model in STAV7.2 in that it does not affect other calculated quantities. An RAI was issued (RAI No. 22 of Reference 2) requesting further justification of this stand-alone feature and the potential impact of hydriding on clad ductility, especially under accident conditions. In response, Westinghouse stated that STAV7.2 is used to assess the impact on fuel performance during normal operations and AOOs and will not be used to assess the dynamic response during postulated transients such as BWR Control Rod Drop. In response to RAI No. 5 (Reference 4), Westinghouse provided details on the hydrogen pickup model in STAV7.2 (which was not presented in WCAP-15836-P). The PNNL technical assessment of the hydrogen pickup model is presented in Section 4.2 of the supporting TER. Based upon comparisons of the STAV7.2 predicted hydrogen content against (1) the Westinghouse SVEA-96 database, (2) an extensive international database, and (3) FRAPCON-3, PNNL concluded that the hydrogen pickup model in STAV7.2 is acceptable.

<u>STAV7.2 Clad Corrosion and Crud Integral Assessment</u>
Figure 3.6-1 of WCAP-15836-P presents a “typical” application of the STAV7.2 corrosion model to recent Westinghouse BWR cladding oxide measurements. Note that the corrosion and crud deposition models use tuning parameters which may vary for different BWR plants and for different Westinghouse BWR cladding types. Based on the data presented and the PNNL

technical assessment, the NRC staff finds the clad oxidation, hydrogen pickup, and crud deposition models capable, for a given database, of predicting clad waterside corrosion and crud deposition. The NRC staff acknowledges that the buildup of corrosion and crud layers may be plant-specific as well as clad-specific. As such, tuning parameters within these corrosion and crud models may need to be adjusted based upon future trends derived from post-irradiation examinations. The application of these models to future licensees will be addressed in WCAP-15942-P.

3.4 Fuel Densification and Swelling Models

PNNL compared the pellet volume change due to densification and swelling in STAV7.2 to FRAPCON-3 (Section 5.0 of supporting TER). Examination of Figure 5-1 of the supporting TER reveals that the maximum densification occurs at a higher burnup and the fuel swelling rate is lower in STAV7.2 relative to FRAPCON-3. A less rapid densification will promote lower fuel temperatures at the beginning of life. A lower swelling rate will promote a lower gap conductance and higher fuel temperatures throughout fuel life.

Each model in a fuel performance code does not need to be independently conservative for every application. It is the interaction of the various models coupled with an appropriate implementation methodology (WCAP-15942-P) which ensures a conservative fuel design. Westinghouse provided a comparison of the STAV7.2 predicted rod void volume to measured data (Figure 3.7-1 of WCAP-15836-P) which shows a best-estimate fit. PNNL concluded that STAV7.2 fuel densification and swelling models are acceptable for thermal analyses and acceptable for evaluating the relative mechanical performance of different fuel designs and operating modes. Based upon the data presented and the PNNL technical assessment, the NRC staff finds the fuel densification and swelling models acceptable. A conservative application of these models will be governed by the mechanical design methodology in WCAP-15942-P.

3.5 Fuel Rod Mechanical Properties

The modeling of mechanical fuel rod behavior in STAV7.2 assumes a rigid pellet (i.e. no fuel creep) and the fuel strain (i.e., fission product swelling, densification, thermal expansion, relocation, and rearrangement) determines the amount of elastic-plastic strain in the cladding when contact between the fuel and the cladding is achieved. The cladding mechanical properties are modeled in STAV7.2 in order to predict the cladding response to in-reactor stresses and predict elastic and plastic deformation and cladding creep.

Since mechanical properties are strongly dependent on cladding types (i.e. alloying and heat treatments), the NRC staff and Westinghouse agreed early in the review to limit the applicability of STAV7.2 to fully recrystallized (RXA) Zircaloy-2 (RAI No. 25, Reference 2).

Most of the mechanical properties in STAV7.2 are identical to those in the previously approved STAV6.2. Those models that have changed are cladding creep, Young’s modulus, and yield strength.

Irradiation Creep

Section 3.4 of WCAP-15836-P (as updated by Attachment 2 of Reference 3) describes the calibration and verification of the STAV7.2 cladding creep model. The database used to calibrate the creep model is primarily in-reactor cladding creep data obtained from tests performed by Babcock & Wilcox in the Oconee nuclear power plant under an Electric Power Research Institute (EPRI) program. Due to the empirical nature of the calibration process, the values of the tuning parameters become inherently part of the approved model. PNNL evaluated the calibration of the creep model (Section 6.2.1 of supporting TER) and concluded that the creep model fit the Oconee data in a best-estimate manner.

Westinghouse's verification of the creep model (RXA Zirc-2) consisted of comparisons to measured creep down from 24 BWR commercial fuel rods and 8 experimental fuel rods irradiated in the Studsvik test reactor. Comparisons presented in Section 3.4.2 of WCAP-15836-P were supplemented with additional creep data in response to RAI No. 21 (Reference 2). Since most of this creep data were based on an earlier version of Westinghouse BWR cladding, an RAI was issued requesting further justification of the creep model to the latest RXA Zirc-2 cladding alloys (e.g., LK2, LK2+, and LK3). In response (RAI No. 3 of Reference 4), Westinghouse provided creep data which showed little difference in creep behavior between the different alloys.

PNNL benchmarked the STAV7.2 creep model against FRAPCON-3. Based upon the verification database and comparisons to FRAPCON-3, PNNL concluded that the creep correlation in STAV7.2 is acceptable for fully RXA Zircaloy-2 cladding. Based upon the calibration and verification exercise presented in Section 3.4 of WCAP-15836-P (and in response to the RAIs) and the PNNL technical assessment, the NRC staff finds the creep model in STAV7.2 acceptable.

High Stress Creep Model

PNNL benchmarked the steady-state creep model in STAV7.2 against the FRAPCON-3 model at BWR conditions and high stress (Section 6.2.2 of supporting TER). Examination of figure 6.2.3 of the supporting TER reveals that the two models predict similar values for creep rate with STAV7.2 predictions slightly more conservative. The Westinghouse database used to verify the creep model was limited to a hoop stress of 100 MPa. PNNL considered whether the model's verification to a limited database would necessitate a code limitation. PNNL concluded that since the application of STAV7.2 will be limited by the no-clad-lift-off criteria (which at approximately 75 MPa for a typical BWR fuel rod is within the 100 MPa database), the creep model is acceptable without a specified limitation due to its supporting database.

Yield Strength and Young's Modulus

STAV7.2 uses Hooke's Law to relate stress and strain in the elastic region and a modified power law to relate stress and strain in the plastic region. PNNL compared the coefficients in STAV7.2 against FRAPCON-3 and mechanical test data and concluded that STAV7.2 strength coefficients were excessively high for irradiated cladding. In response to RAI No. 8 (Reference 4), Westinghouse provided further justification for the strength coefficients in STAV7.2. As part of their response, Westinghouse agreed to better treat the strength coefficient at high temperatures. The STAV7.2 yield strength correlation was modified to include a correction factor, which is a function of the liner thickness.
The yield stress used in STAV7.2 will be the larger of either the STAV7.2 predicted yield stress (adjusted by correction factor) or the yield stress predicted by the original STAV6.2 model.

Because the STAV6.2 correlation does not contain a term to correct for the thickness of a clad liner, applicability of STAV7.2 will be limited to fuel rod designs with a maximum clad liner thickness of 4 mils (nominal) which is the upper extent of the database used in the calibration and verification of the yield stress model.

PNNL compared the revised STAV7.2 yield stress model to FRAPCON-3 for both unirradiated and irradiated cladding. Examination of Figure 6.2.4 through 6.2.6 reveal that the behavior of both models (with fluence and temperature) are similar in shape and magnitude. Neither PNNL nor Westinghouse was able to produce data showing yield stress above 698K for RXA Zirc-2 cladding. As a result of this lack of data within the industry, the applicability of STAV7.2 will be limited to applications with cladding average temperature at any axial node less than 698K. Based on the data presented in WCAP-15836-P and in response to RAIs and the comparison to FRAPCON-3, PNNL concluded that the yield stress model is acceptable.

Up to the applicability limit of 698K, the model used for Young's modulus in STAV7.2 is identical to the model in FRAPCON-3. PNNL concluded that this is acceptable.

Based upon the data presented in WCAP-15836-P and in response to RAIs and the PNNL technical assessment, the NRC staff finds the revised yield stress model acceptable up to a clad average temperature of 698K (425EC) at any axial node and a clad liner thickness of 4 mils (nominal).

<u>Pellet-Cladding Mechanical Interaction</u>
Section 3.8.2 of WCAP-15836-P (as updated by Attachment 2 of Reference 3) describes the verification of the STAV7.2 PCMI model. PNNL assessed the pellet-cladding mechanical interaction in STAV7.2 against Halden test IFA-404.1. This Halden rod was ramped up from zero power to approximately 50 kW/m and then back down to zero power. Instrumentation measured axial and hoop strains during the power ramps. The STAV7.2 prediction of hoop strain was compared to the Halden data and found to be in reasonable agreement. PNNL concluded that the STAV7.2 predictions of hoop and axial strain for this case are acceptable. Based on the PNNL technical assessment, the NRC staff finds the PCMI modeling in STAV7.2 acceptable.

3.6 Void Volume

The void volume in STAV7.2 is calculated in the same manner as in STAV6.2. The predicted void volume is affected by many of the models in STAV7.2, including fuel pellet swelling, densification, cladding creep, fuel rod growth, fission gas release, and temperature distribution.

<u>Fuel Rod Growth</u>
The model for fuel rod growth in STAV7.2 for BWR cladding is the same as the model in STAV6.2. PNNL compared the PWR and BWR fuel growth models in STAV7.2 against FRAPCON-3 for Zircaloy-2 (BWR) and Zircaloy-4 (PWR). Note that the FRAPCON-3 models are based on an EPRI model and has been validated to a local burnup of 65 GWd/MTU. Examination of Figure 7.1.1 in the supporting TER reveals good agreement between the models in STAV7.2 and FRAPCON-3. The Westinghouse database employed to validate STAV7.2 consists of BWR fuel data beyond 62 GWd/MTU. Based on the data comparisons, the PNNL technical assessment concluded that the STAV7.2 model for BWR fuel rod growth is acceptable. Based upon the data presented in WCAP-15836-P (and Figure 4.2-7 of

WCAP-15942-P, currently under review) and the PNNL technical assessment, the NRC staff finds the STAV7.2 model for BWR fuel rod growth acceptable up to 62 GWd/MTU.

Plenum Gas Temperature

The plenum gas temperature in STAV7.2 is set equal to the coolant temperature at the axial node of the plenum. PNNL recognized that this assumption is slightly non-conservative because some heat will be transferred into the plenum from the end pellet and some heat will be produced in the plenum by gamma heating in the plenum spring (Section 7.2 of supporting TER). In response to RAI No. 6 (Reference 4), Westinghouse agrees that this simplistic modeling assumption is non-conservative, especially for the plenum in the part length rod. However, Westinghouse stated that "this effect on hot gas pressure is within the noise of the ability to calculate pressure and insignificant relative to the uncertainties that are taken into account." PNNL agrees with these assessments and concluded that the effects of this slight non-conservatism will be minor. The NRC staff agrees with the PNNL technical assessment of the plenum gas modeling in STAV7.2 and finds it acceptable.

Void Volume Integral Assessment

Section 3.7 of WCAP-15836-P (as updated by Attachment 2 of Reference 3) describes the verification of the STAV7.2 void volume calculations. The PNNL integral assessment of the void volume calculations is discussed in Section 7.0 of the supporting TER.

The Westinghouse database, utilized in the verification process, is a subset of the database used in the calibration and verification of the steady-state fission gas release models. Of this database, hot cell void volume measurements were made on 66 BWR and 129 PWR fuel rods. Figure 3.7.1 (as updated by Attachment 2 of Reference 3) illustrates measured versus predicted void volumes at room temperature. The BWR and PWR database supporting the void volume calculations has a maximum burnup of 52 GWd/MTU and 61 GWd/MTU, respectively.

Based on the data comparisons, the PNNL technical assessment states that the STAV7.2 code predicts void volume in a best-estimate manner with a relatively small degree of scatter. PNNL concluded that the STAV7.2 void volume model is acceptable. Based upon the data presented in WCAP-15836-P and the PNNL technical assessment, the NRC staff finds the void volume calculation acceptable.

3.7 Licensing Applications

In order to investigate the synergistic interaction of coupled models and ensure a conservative licensing application of STAV7.2, Westinghouse provided sample fuel rod design applications of a typical BWR fuel assembly and PNNL benchmarked these licensing applications to FRAPCON-3. The fuel mechanical design methodology, including the application of uncertainties, will be addressed in WCAP-15942-P. The scope of this investigation is to assess the ability of STAV7.2 to predict fuel rod performance in a best estimate or conservative manner depending on the analyses.

Fuel Melting

PNNL benchmarked the fuel melting temperature in STAV7.2 against FRAPCON-3 (Section 8.1 of the supporting TER). In addition to a comparison of steady-state fuel melting temperature with burnup, fuel temperature benchmark cases, including six segmented power histories (SPHs) and an anticipated operational occurrence (AOO) for full length and part length UO_2 rods, were performed. PNNL concluded that for the fuel melt analysis, STAV7.2 predicts conservative values for fuel melting temperature and conservative values for maximum fuel temperature for a rod average burnup up to 62 GWd/MTU. Based upon the PNNL technical assessment, the NRC staff finds the application of STAV7.2 to the fuel melting analysis acceptable.

Fuel Stored Energy

PNNL benchmarked the peak node centerline temperature in STAV7.2 against FRAPCON-3 for two different power histories (Section 8.2 of supporting TER). For both cases, STAV7.2 predictions of peak node centerline temperature were conservatively higher than FRAPCON-3. PNNL concluded that STAV7.2 is acceptable for LOCA initialization (input to CHACHA-3) at reasonable power level and temperature, but may be excessively conservative at high power levels and temperatures due to conservative fission gas release predictions. Based upon the PNNL technical assessment, the NRC staff finds the application of STAV7.2 to the LOCA stored energy analysis (via input to CHACHA-3) acceptable.

Fuel Rod Internal Pressure

PNNL benchmarked the rod internal pressure in STAV7.2 against FRAPCON-3 over a range of power levels including the six SPH and several different AOO power pulse scenarios for full length and part length fuel rods (Section 8.3 of supporting TER). In each case, STAV7.2 predicted conservatively higher rod internal pressures than FRAPCON-3. PNNL found that the difference between the two codes' prediction of rod internal pressure is due to the conservatively high fission gas release predicted by STAV7.2. PNNL concluded that the STAV7.2 code is acceptable for application to fuel rod pressure analysis. Based on the PNNL technical assessment, the NRC staff finds the application of STAV7.2 to the rod internal pressure analysis acceptable.

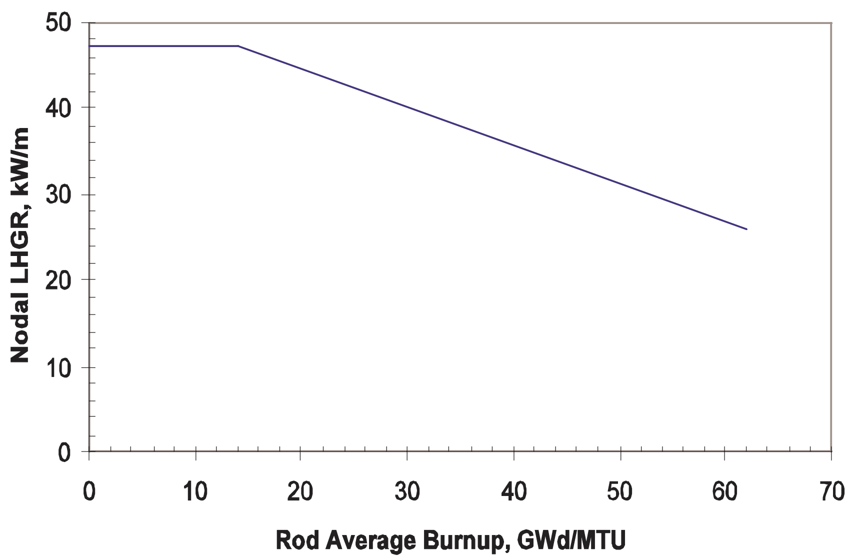
Clad Strain

PNNL benchmarked the maximum cladding strain in STAV7.2 against FRAPCON-3 for a limiting AOO case including three power pulses each lasting thirty hours (Section 8.4 of supporting TER). The maximum hoop strain predicted by STAV7.2 was greater than that predicted by FRAPCON-3. PNNL determined that the reason FRAPCON-3 predicts less strain than STAV7.2 is because no stress is transferred to the cladding while the relocation of the pellet is being taken up in FRAPCON-3. While in STAV7.2, a condition of soft contact is defined where some stress is transferred to the cladding while the relocation is being taken up. PNNL concluded that the STAV7.2 code is acceptable for application to cladding strain analyses. Based on the PNNL technical assessment, the NRC staff finds the application of STAV7.2 to cladding strain analyses acceptable.

3.8 STAV7.2 LHGR Limitation

Longer cycle lengths and higher power core reload designs (e.g. Extended Power Uprate) promote more aggressive fuel utilization with higher fuel rod power throughout fuel rod burnup. It is important to verify the applicability of the STAV7.2 fuel performance models (and supporting fuel experience database) to these higher power cores. In response to an RAI (Reference 5), Westinghouse provided a proposed LHGR limit as a function of rod average burnup along with the power histories for the fuel rods in their fission gas release and fuel temperature database. Figures 3 through 6 of Reference 5 provide an investigation of the STAV7.2 model predictions versus measurements taken from fuel rods with power histories adjacent to the proposed LHGR limit. In Section 8.5 of the supporting TER, PNNL concluded that the calibration and verification data used in the development of STAV7.2 support operation at or below the LHGR limit shown below. It should be noted that this limit applies only to steady-state LHGR and does not apply to transient LHGR such as for an AOO. The peak LHGR during a transient may exceed this LHGR limit for the short duration of the transient but must meet the LHGR versus time duration used for analyzing AOO events. Based upon the information presented in Reference 5 along with the PNNL technical assessment, the NRC staff finds that STAV7.2 is applicable up to the nodal power profile depicted below.

The figure below illustrates the peak LHGR $\leq$ 47.3 kW/m from BOL to a rod average burnup of 14 GWd/MTU, then linearly decreasing to 26 kW/m at a rod average burnup of 62 GWd/MTU.



3.9 VIK-3

The computer code VIK-3, described in Section 4 of WCAP-15836-P, performs cladding stress and end plug weld area stress analyses. A majority of the code features remain unchanged from VIK-2, approved in CENPD-285-P-A (Reference 6). The PNNL technical assessment of VIK-3 is provided in Section 9 of the supporting TER.

In VIK-3, stress calculations can be performed as a function of fuel rod burnup using STAV7.2 material properties, fuel rod parameter inputs, and loads. This code feature allows the code a greater degree of integration with STAV7.2.

The Paidoussis correlation has been added to VIK-3 which provides amplitudes due to flow induced forces to the rod bending calculation. In response to RAI No. 1 (Reference 2) and RAI No. 7 (Reference 4), Westinghouse provided a comparison of the VIK-3 predictions against the experimental data used to validate the code.

The finite difference technique used in VIK-2 to calculate the stress and temperature distribution in the bottom end plug has been replaced with a finite element technique in VIK-3. In response to RAI No.3 (Reference 2), Westinghouse provided a more detailed description of the finite element code and a typical finite element mesh.

Based upon their review of the information presented in WCAP-15836-P and in response to RAIs, PNNL concluded that the VIK-3 code is acceptable for cladding stress analysis and end plug weld area stress analyses. Based upon the PNNL technical assessment, the NRC staff finds the VIK-3 code acceptable.

3.10 COLLAPS-3.3D

The computer code COLLAPS-3.3D, described in Section 6 of WCAP-15836-P, calculates cladding ovalization due to creep, up to the point of mechanical instability and creep collapse. A majority of the code features remain unchanged from COLLAPS-3.2S, approved in CENPD-285-P-A (Reference 6). The PNNL technical assessment of COLLAPS-3.3D is provided in Section 10 of the supporting TER.

Changes incorporated into COLLAPS-3.3D are listed below:

- Use of double precision to increase the computational accuracy of the code.

- Optional correction to the infinitely long solution to account for the effect of the pellet support provided at the ends of a finite length pellet-to-pellet axial gap.

- STAV7.2 creep correlation for fully-annealed cladding.

In response to RAI No. 1 and No. 2 (Reference 2), Westinghouse provided background information on the Studsvik data used to validate the creep model. In response to RAI No. 3 (Reference 2), Westinghouse provided background information on the collapse data used to verify the COLLAPS-3.3D model. In response to RAI No. 4 (Reference 2), Westinghouse provided further discussion on the determination of the finite gap length.

Based upon the information presented in WCAP-15836 and in response to RAIs, PNNL concluded that the COLLAPS-3.3D code is acceptable for calculating cladding ovality due to creep, up to the point of mechanical instability and creep collapse. Based upon the PNNL technical assessment, the NRC staff finds the COLLAPS-3.3D code acceptable.

4.0 CONCLUSION

Based upon its review of this TR and technical support provided by the PNNL, the NRC staff finds Westinghouse's suite of BWR fuel performance computer models, STAV7.2, VIK-3, and COLLAPS II Version 3.3D, acceptable. Licensees referencing this TR will need to comply with the conditions and limitations listed in Section 5.0.

5.0 CONDITIONS AND LIMITATIONS

Licensees referencing WCAP-15836-P must ensure compliance with the following conditions and limitations:

1. STAV7.2 is approved for modeling BWR fuel rods with the following limitations.

    a. Solid UO_2 fuel pellet with a maximum gadolinia content of 9.0 wt%.
    [Requested by Westinghouse, see Section 3.1]
    b. No substance beyond gadolinia and nominal trace elements shall be added to the fuel pellet for the purposes of altering its physical characteristics.
    [Fuel additives not part of review scope, see Section 3.1]
    c. Nominal fuel pellet density between 92 - 97 percent theoretical.
    [Extent of fuel thermal conductivity database, see Section 3.1]
    d. Fully RXA Zircaloy-2 fuel clad material.
    [Clad properties calibrated to RXA Zircaloy-2 database, see Section 3.5]
    e. For fuel rods with a clad liner (e.g. natural zirconium), the liner thickness shall be no greater than 4 mils (nominal).
    [Extent of cladding yield stress database, see Section 3.5]
    f. Peak rod average burnup limit of 62 GWd/MTU.
    [Requested by Westinghouse, see Section 1.0]

2. STAV7.2 shall not be used to model fuel above incipient fuel melting temperatures.
[Limitation on fuel thermal expansion model, see Section 3.1]

3. STAV7.2 shall not be used to model fuel rods with an average cladding temperature above 698K (425 EC) at any axial node.
[Extent of cladding yield stress database, see Section 3.5]

4. STAV7.2 shall be used only within the range for which fuel performance data were acceptable or for which the verifications discussed in WCAP-15836-P and responses to RAIs were performed. For example, Section 3.8 describes a LGHR limit based upon the calibration and verification database of STAV7.2.
[Applicability of STAV7.2 shall remain within NRC scope of review and acceptance, see Section 3]

5. Due to the empirical nature of the STAV7.2 calibration and validation process, the specific values of the equation constants and tuning parameters derived in WCAP-15836-P (as updated by RAIs, e.g. Attachment 2 of Reference 3) become inherently part of the approved models. Thus, these values may not be updated without further NRC review. Exceptions include the BWR cladding corrosion constants (Table 2.2.51), crud deposition constants (Table 2.2.5-2), and rod nodal power uncertainties for the BWR "Older" data (Uncontrolled and Controlled Cells in Table 3.3-1). These exceptions will be addressed as part of the implementation methodology in WCAP-15942-P.
[Applicability of STAV7.2 shall remain within NRC scope of review and acceptance, see Section 3]

6.0 REFERENCES

1. Letter from I. C. Rickard (Westinghouse) to U.S. Nuclear Regulatory Commission, "Submittal of Topical Report WCAP-15836-P, Revision 0, 'Fuel Rod Design Methods for Boiling Water Reactors - Supplement 1'," LTR-NRC-02-29, June 25, 2002, ADAMS Accession No. ML021780297.

2. Letter from J. A. Gresham (Westinghouse) to U.S. Nuclear Regulatory Commission, "Part Response to NRC Request for Additional Information (RAI) on WCAP-15836-P, Revision 0, 'Fuel Rod Design Methods for Boiling Water Reactors - Supplement 1'," LTR-NRC-04-22, April 16, 2004, ADAMS Accession No. ML041110704.

3. Letter from J. S. Galembush (Westinghouse) to U.S. Nuclear Regulatory Commission, "Response to NRC Request for Additional Information (RAI) on WCAP-15836-P, Revision 0, 'Fuel Rod Design Methods for Boiling Water Reactors - Supplement 1'," LTR-NRC-04-41, July 30, 2004, ADAMS Accession No. ML042240440.

4. Letter from J. A. Gresham (Westinghouse) to U.S. Nuclear Regulatory Commission, "Responses to NRC Request for Additional Information on WCAP-15836-P, 'Fuel Rod Design Methods for Boiling Water Reactors - Supplement 1' (Proprietary/Non-Proprietary), dated March 2005, TAC No. MB5740," LTR-NRC-05-10, March 9, 2005, ADAMS Accession No. ML050730315.

5. Letter from J. A. Gresham (Westinghouse) to U.S. Nuclear Regulatory Commission, "Responses to NRC Request for Additional Information on WCAP-15836-P, 'Fuel Rod Design Methods for Boiling Water Reactors - Supplement 1' (Proprietary/Non-Proprietary), dated April 2005, TAC No. MB5740," LTR-NRC-05-29, April 22, 2005, ADAMS Accession No. ML051250610.

6. CENPD-285-P-A, "Fuel Rod Design Methods for Boiling Water Reactors," July 1996, ADAMS Accession No. 9610310097.

Principal Contributor: Paul Clifford, NRR

Date: June 2, 2005